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Abstract

IRRADIATION BEHAVIOUR OF COATED FUEL PARTICLES

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In order to obtain information on the behaviour of coated fuel particles under reactor conditions a series of capsule tests was performed in the ASTRA-reactor at thermal flux levels of $3 - 6 \times 10^{13}$ n/cm^2 sec. The burn-up achieved was about 10% fifa at operating temperatures of 1200 - 1400°C.

For these burn-up experiments two types of capsules have been used. One employed a graphite insert which was thermally insulated against the stainless steel containment by a helium gap. The other type utilized a niobium insert thermally insulated by Al_2O_3 powder. Both inserts contained the particles in an annular gap. The fuel temperature was monitored by high temperature thermocouples located at both sides of the fuelled annulus. With this arrangement data of the thermal conductivity of a loose packing of coated particles were obtained.

After dismantling the capsules in the hot cell the burn-up was determined by evaluation of cobalt dosimeters and by mass-spectrometric analysis of the uranium isotopes. This measurement was performed by using a high temperature ion source in which the uranium ions were obtained by surface ionization after a volatilization by a halogen treatment.

After a visual inspection and a metallographical examination the integrity of the particles was checked by two methods. The first utilized a zenon release measurement on reactivated particles to obtain the number of failed particles. The second used a chlorination process which allows the quantitative removal of uranium from failed particles via the volatile chloride.

By analysing the insert of the capsule the amount of non-gaseous fission products released during the experiments was determined by radiochemical methods and in some cases by gamma-spectrometry.

The results of the high temperature burn-up tests are given and correlated with the results of a preirradiation evaluation of coated particles produced by different methods.

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BURN-UP DETERMINATION OF NUCLEAR FUEL BY HIGH RESOLUTION GAMMA-SPECTROSCOPY

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The γ-radiation of fission products is used for burn-up determination of irradiated fuel-elements by a non-destructive method. If no information on the irradiation history (except an upper limit of a few years) is available or cannot be used for the interpretation of the measurements (a case which occurs practically during the inspection of IAEA-distributed fuel) only Cs^{137} can be used for total burn-up determination due to its 30 γ half-life. In addition a measurement of the 2.19 MeV line of Ce-Pr-144 (284 d half-life) is useful due to the low absorption corrections for this γ-energy and favourable diffusion behaviour of these fission products but the interpretation requires a knowledge of irradiation history. The measurement of Cs^{137} in the fission product spectrum is very difficult due to strong adjacent lines, and requires high γ-energy resolution. The measurement of the 2.19 MeV line of Pr-144 is also complicated in Pu-containing fuel due to neighbouring lines of Rh-106.

A high resolution γ-spectrometer is described and practical results are reported. The γ-spectrometer is based on the measurement of the Compton-scattering events in a Li-drifted semi-conductor detector. The principle of the Compton coincidence spectrometer - known since many years for scintillation detectors - is applied. This gives the advantage of a unique correspondence between γ-energy and the observed peak in the pulse-height distribution. By using the high resolution properties of a semi-conductor detector a f.w.h.m. of 18 keV nearly independent on the primary γ-energy has been achieved and further improvement is likely. For the quantitative determination of Cs^{137} a minimum cooling time of two weeks is

required. The spectrometer can be constructed in a very compact way (40 x 40 x 20 cm without electronics and auxiliary apparatus) and requires at the location of the fuel elements storage a collimator hole in the storage tank wall and a scanning machine for the fuel element itself.

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Abstract

RESONANCE ABSORPTION IN HTGR-FUEL ELEMENTS CONTAINING FERTILE AND FISSILE MATERIAL AS COATED PARTICLES IMBEDDED IN A GRAPHITE MATRIX

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In most HTGR cores the fuel elements contain fertile and fissile material as carbide particles, coated with pyrolytic carbon and imbedded in a graphite matrix of particular shape. The distribution of fertile and fissile material in the fuel elements and in the core affects the most important core parameters, such as resonance capture and conversion ratio.

The most accurate analytical method to date of calculating resonance capture is the direct numerical solution of the integral equation for the mean flux in the absorber.

L.W. Nordheim has developed a method for treating the grain structure of fuel elements taking into account the macroscopic heterogeneities. The basic idea is to take the grains themselves as the fundamental elements. Following this way, it is investigated in detail how the different distributions in the core of fertile and fissile materials can be treated with this method and how they affect the basic core parameters.

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Abstract

ANALYTICAL METHODS IN ENERGY DEPENDENT TRANSPORT THEORY

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The time dependent neutron density in six-dimensional configuration space is of fundamental importance for the computation of the various parameters and functions describing a reactor. It is solution of the Boltzmann equation with special boundary and initial values that are determined by the reactor under consideration. The collision operator on the right side of the equation depends only in the collision and transport properties of the reactor medium. They also can depend in space and time:

$$\int d\vec{v}' \; K(\vec{v}' - \vec{v}, \vec{x}, t) - 1^{-1}(\vec{x},\vec{v},t)$$

There is a variety of approximation methods. They make possible numerical calculations with more or less accuracy and security. Concerning the existence and structure of the solutions there are only a few papers (1958, G.M. Wing). Using real and complex analysis they give complete and rigorous solutions of the energy in dependent Boltzmann equation and demonstrate that some folklore common to textbooks in reactor theory turns to be false. The operators appearing in the energy dependent Boltzmann equation or in derived equations are not selfadjoint. These operators are more general than the observables usually used in quantum-mechanics. This fact complicates the solution of the problem. Using the theory of linear operators in Hilbert space two major examples are considered in detail:

(1) The stationary neutron spectrum in an unbounded reactor and

(2) The time dependence of the neutron density in a slab reactor.

The basis models of the computation are very realistic. No mathematical approximations are made. So one can expect that good approximation methods can be derived with an arbitrarily given degree in accuracy in contrast to all other methods.